REPORT of Oct. 7-8 MEETING of the FFCC

A meeting of the Fusion Facilities Coordinating Committee was held at MIT on October 7-8. Since this was the first meeting of the Committee, it addressed issues of process as well as issues concerned with the charter to the committee. This report will summarize the Committee's discussions, recommendations, and conclusions. The four major topics which were addressed at the meeting were: Program Plans for DIII-D, C-MOD and NSTX; status of tokamak and ST strategic white papers; research objectives for FY'99 and FY'00; and coordination among the facilities.

FACILITY PROGRAM PLANS

DIII-D PROGRAM PLANS

by T. C. Simonen

This year the DIII-D program embarked on a new five year program with the mission to establish the scientific basis for optimization of the tokamak approach to fusion energy production. DIII-D's broad program goals are to advance the science of magnetic confinement on a broad front, to seek to find the ultimate potential of the tokamak as a magnetic confinement system, and to resolve key enabling issues for advancing various magnetic fusion concepts.

This program stresses science understanding and has a high priority objective to establish the viability of the advanced tokamak concept. Today's high performance advanced tokamak experiments without ECCD are limited in duration to one second due to plasma current profile evolution and the onset of the neoclassical tearing mode. On a three-year time scale, with 6 MW of 110 GHz ECH power, we aim at an integrated demonstration of sustained advanced tokamak operation for five seconds. On a five year horizon, with 10 MW of ECH power and improvements in the magnet power system, the DIII-D AT experiments would be extended to higher betas and to 10 second duration, as shown in Fig. 1.

The three-year DIII-D facility plan is shown in Fig. 1. The facility is now undergoing annual maintenance and diagnostic instrument calibration and improvements. We are installing a third 1 MW gyrotron, guide tubes for inside launch pellet injection, and a central Thomson scattering system. Vacuum vessel close is scheduled for late October, leak checking and vacuum conditioning during November, plasma checkout in December, and physics experiments will begin on January 11 continuing until August. Eighteen weeks of physics experiments are scheduled.

During FY99 construction will begin on three additional megawatt 110 GHz gyrotron systems increasing the ECH power during year 2000 to 6 MW. A PPPL power supply will be installed in the spring of 1999 in order to carry out initial high beta feedback stabilization experiments in the summer. A divertor pump and baffle will be fabricated for installation in the fall of 1999. Physics experiments would begin in February 2000.

Approximately two hundred proposals were presented by DIII-D researchers at a September 22-24 "Brainstorming Meeting." These ideas were related to research programmatic thrusts identified by the Research Council and to specific physics topical areas. The central theme of the 1999 research is Advanced Tokamak High Beta Stability. This program will be carried out with research thrusts dealing with regulating edge bootstrap currents, stabilization of neoclassical tearing modes, resistive wall mode stabilization, single vs. double null, and internal transport barriers. These thrusts will be coupled to experiments preparatory for the sustained advanced tokamak demonstration in 2001 (optimized shear and high internal inductance target discharges). A number of basic fusion science experiments will also be carried out, many of which provide desired complementary data to Alcator C-Mod research and with international tokamaks, JET, JT-60U, ASDEX-Upgrade, and TEXTOR.

ALCATOR C-MOD PROGRAM PLANS
by I. H. Hutchinson

The Alcator C-Mod program begins a new five-year program in January 1999, focused on compact high-performance tokamak research to establish the plasma physics and plasma engineering for an ignited tokamak experiment and attractive fusion reactors. The program emphasizes the exploitation of the unique capabilities and features of Alcator C-Mod, the world's highest magnetic field tokamak.

Major upgrades are being completed in support of the research objectives. The ICRF heating and current drive power has been doubled by the installation of a new ICRF launcher developed in collaboration with PPPL and optimized for Mode-Conversion Current Drive. This installation, in addition to permitting higher plasma temperatures and betas to be obtained, will enable critical research to begin on the control of the current profile for confinement and stability optimization, and on the control of plasma flow, responsible for turbulence suppression. A modification is being installed to the divertor baffles allowing dynamic control of the recirculation of neutral gas from the divertor to the main chamber. This facility, called the “flaps”, allows the effects of neutrals on the H-mode edge pedestal to be explored as well as the neutral particle role in the divertor itself in such important processes as detachment, momentum removal, and divertor power and particle handling. A diagnostic neutral beam, installed in collaboration with U Texas, will begin operation in the FY99 campaign, permitting a major enhancement of the diagnosis of ion temperature and velocity profiles and of the magnetic field structure. These and other diagnostic upgrades, aimed at the scrape-off-layer and edge pedestal, pave the way for detailed measurements in support of transport and stability research.

Upgrades in succeeding years include expanding the plasma current capability of the tokamak through upgrade of the internal components, the installation of a divertor “septum” for additional divertor neutral control, and the installation of a cryopump, as outlined in the schedule of figure ... Alcator proposes in the longer term to pursue quasi-steady-state Advanced Tokamak research through the installation of the refurbished Lower Hybrid system for current drive and current profile control. However engineering of this system cannot begin in FY99 on the present budget or in FY’00 on a flat budget.

The present program plan is for 12 weeks of physics operations in FY99 (18 weeks in FY’00 and FY’01). The FY99 campaign has been planned on the basis of an ideas forum, held on 19-20 Aug. 98 (web-broadcast on the “Mbone”) at which 118 ideas were presented by 50 speakers, representing 8 different institutions. This was followed up by detailed prioritization by 14 open “focus groups” and subsequently organization into five themes: Neutral interactions and divertor physics, Plasma rotation and transport, RF current- and flow-drive and energetic particles, H-mode pedestal physics and ELM behavior, and Plasma performance and facility development.

Alcator C-Mod provides critical data on plasma behavior in a unique parameter range that complements research on larger facilities such as DIII-D and the international tokamaks, directed towards basic magnetic confinement science and towards establishing advanced tokamak performance. In addition C-Mod supports the high-field compact tokamak route to fusion ignition, which appears an attractive way to address the burning plasma fusion issues in the near-term.

NSTX RESEARCH PROGRAM
By Martin Peng

The experimental phase of NSTX is planned to begin in May 1999. Other new ST experiments worldwide (MAST/U.K., Pegasus/U.S., Globus-M/R.F., and ETE/Brazil), with varying research capabilities and emphases, are being built to achieve first plasma near the close of 1998 and join NSTX, HIT-II, and CDX-U in ST research.

The mission of the NSTX Research Program is to prove the physics principles of the following attractive ST properties:

- Plasma startup without complicated inboard solenoid magnet,
- Efficient heating and non-inductive current drive for steady-state operation,
- High plasma pressure in low magnetic field for high fusion power density and low cost,
- Nearly fully self-driven (bootstrap plus diamagnetic) plasma current for economic operation,
- Good energy confinement to permit small-size fusion plasma, and
- Dispersed heat and particle fluxes.

The NSTX Research Program is envisioned to encompass three experimental phases through the next 5 years: I) Startup and Ohmic Heating; II) First Stability Regime; and III) Advanced Physics Regime. FY99-01 will be devoted to the first two phases. Phase I will afford the first opportunity in 15 research run weeks to test Ohmic operation, CHI startup, and injection of modest levels of HHFW power. The First Stability Regime (Phase II) is expected to afford 21 research run weeks and investigate plasmas characterized by average toroidal betas up to 25% for q ~ 10 with significant bootstrap current fraction ~ 50%, without mode control, and confinement at par with those indicated by the tokamak scaling expressions (such as the ITER power law 89P). Detailed control of plasma profiles is not anticipated to be necessary in this regime. The Advanced Physics Regimes (Phase III) plasmas are expected to be characterized by average toroidal betas up to 40-50% for q ~ 10-15 with mode control, nearly full bootstrap current fraction ~ 80-90% well aligned with the total current profile, and confinement approaching the level of neoclassical ions. Detailed control of all plasma profiles is anticipated to be necessary to reach this regime.

Confirmation of the First Stability Regime physics would enable viable steady state volume neutron source designs that are modest in size and fusion power while delivering high neutron wall loading. The Advanced Physics Regime would be required for economy of future fusion power plants.

The NSTX Facility, together with other contemporary ST experiments, will enable investigations in significant pulse lengths (up to 5 s) using RF and NBI plasma heating and diagnostics techniques already available at PPPL and to be provided by collaborating researchers. The NSTX Facility plan is summarized in Fig. 1. The first plasma is expected in February 1999 whilst the completion of the NSTX Project is expected by end of April 1999. Commissioning and testing is to be completed for experiments to begin in July 1999. A multi-time and point laser Thomson scattering system is expected to be available in August 1999. A spare TFTR NBI System will be refurbished for installation to begin in January 2000 to enable combined high-power HHFW and NBI heating and current drive experiments to begin in July 2000.

A collaborative national team is being formed by DOE and will carry out the NSTX Research Program. A management approach is being developed with DOE to implement the national research program. Desirable upgrades are being incorporated into the NSTX National Research Plan. A NSTX Program Advisory Committee, composed of senior U.S. and world fusion researchers, has been active since July 1996 in shaping the so-far very successful NSTX Project and Program.

INTERNATIONAL COLLABORATIONS PROGRAM PLANS

by N. R. Sauthoff

The U.S. program in international collaborations on fusion continues with a focus on the achievement of U.S. fusion program goals via international collaboration. The international program should be integrated with and be complementary to the domestic program. The motivation for international collaborations involves access to plasma conditions and device configurations that are unavailable domestically and extension of US discoveries and innovations to other plasma conditions and international programs. The U.S. program is refining its implementation of collaborations via a community working group led by PPPL at DOE's request, and is basing its direction on the DOE Strategic Plan for International Collaborations in Fusion Research. The working group is giving highest priority to the achievement of US fusion program goals and to the intents of the programs and program managers of the international facilities. The group is developing recommendations in the areas of tokamaks, innovative/alternate concepts, theory, inertial fusion energy, and technology. In the magnetic fusion area, the topical areas are transport and turbulence, MHD equilibrium and stability, wave-particle interactions, boundary/edge physics, energetic particle effects, and integration. In inertial fusion energy, the topical areas include target physics and heavy ion drivers. In technology, they include materials, power and particle handling, blankets, magnets, tritium handling, safety, and remote handling.

Collaborations currently exist in a broad range of areas on facilities worldwide; control of transport barriers and analysis of the evolution of turbulence are pursued in devices including JT-60U (Japan), JET (Europe), and ASDEX (Germany); current profile measurements and studies of the effects of profile modification are underway on JET and JT-60U; studies of the size-scaling of transport barriers and stability are in place on JET and JT-60U, studies of the RI mode are underway on TEXTOR (Germany); and US physicists participate in all 7 of the ITER Physics Expert Groups. The community is assessing further opportunities on existing facilities and the benefits of US participation in future programs, such as KSTAR (Korea), HT7-U (China), and IGNITOR (Italy).

In innovative/alternate concepts, the US is studying size-scaling in LHD (Japan), and the effects of the varying stellarator configurations (transform, shear, and helical axis shift) in LHD, W7-AS (Germany), and TJ-II (Spain). The US is extending its domestic program in spherical toruses by work on MAST (UK), Globus-M (Russian Federation), TS-4 (Japan), and ETE (Brazil). The US RFP program is extending to RFX (Italy), TPE-RX (Japan), and T2 (Sweden).

In IFE, target physics work is pursued on the Gekko facility (Japan) and GSI (Germany). Heavy ion driver work is extended to GSI. Laser driver work involves the Laser Magajoule facility (France), and studies of diode-pumped solid state lasers in Japan.

In technology research, the materials irradiation studies benefit from use of reactor facilities abroad, such as the BOR-60 reactor in Russia. Heating, current drive and fueling work involves JET and may be extended on JET, KSTAR and other facilities. Magnet work is mostly in the Central Solenoid Model Coil studies, where the US was a major partner (with Japan) and the testing is beginning in Japan.

The US government has decided to end US participation in ITER. A new structure and agreement for international collaborations is planned for the Spring of 1999.

STATUS of STRATEGIC WHITE PAPERS for TOKAMAK and ST RESEARCH

The existing strategic white papers for tokamaks and STs were discussed. These documents could provide the framework for the committee to discuss the contributions from both programs in the near-term. Due to changes in the status of ITER, it was felt that the tokamak paper needed to be revised. In addition, it addressed what the tokamak program was seeking to accomplish in general terms but did not lay out what is needed to be accomplished during the next five to ten year period to advance the concept. This document could also be a useful input to both SEAB and to the Fusion Summer Study next summer which will discuss the future of the tokamak program. One of the outstanding issues which was not discussed is whether we should plan on a successor facility to DIII-D or C-MOD in the U.S. during the next decade or plan on relying on international collaborations. Tom Simonen will take the lead in revising the tokamak white paper.

The ST white paper was more of a roadmap for the development of the ST concept and emphasized long term applications. It also did not lay out what the ST program needed to accomplish during the next five to ten year period. This document should also be updated with less emphasis on long term applications. Martin Peng will take the lead in revising it.

To be available for the SEAB review, revised documents are needed by the end of December.

RESEARCH GOALS for FY99 and FY00

Prior to the FFCC meeting each of the three major facilities identified five key persons to develop a list of research goals for FY99 and FY00 in the science topical areas of transport, stability, power and particle exhaust, wave particle interaction, and composite and integration physics. The five inputs (attached as Appendix A) are very detailed and technical. To convey a general sense of prioritization and importance of the research, especially to readers who were not technical experts in the topical area, members of the FFCC undertook to identify topics of broad scientific interest and express the goals in general terms. These goals are attached as Appendix B. They are being iterated with the individual programs and their key persons and will be discussed with their respective program advisory committees in February and finalized at the February 25-26 FFCC meeting prior to preparing field work proposals. Both Appendix A and B should be treated as draft documents which have not been reviewed by the Committee.

FACILITY COORDINATION

Near the end of the meeting, we briefly discussed the coordination of activities across the facilities. In addition to the FFCC, several mechanisms are already in place to facilitate coordination. Scientists from each facility attend the other's brainstorming meetings and are members on the PACs. In recent years, the ITER Expert Groups have provided a mechanism for scientific exchange and coordination of experiments. In addition, preparation for this meeting and of the topical white papers resulted in increased coordination and discussion.

Several examples of coordination of activities were identified in the discussion.

- The DIII-D group has de-emphasized research on "transformerless operation" since this is a major thrust area of the NSTX device.
- Though C-Mod and DIII-D run schedules overlap during the winter and spring of FY'99, the NSTX operations will occur during the summer facilitating the collaborations during the startup of experiments on NSTX. Finer scale coordination usually occurs to accommodate schedules and needs of individual collaborators. In general, the committee did not believe that it was practical to coordinate in detail the operating schedules since unplanned events make such coordination very difficult.

- In RF physics, DIII-D will take the lead in ECH/ECCD whereas Alcator C-Mod in ICRF minority and mode conversion heating and current drive and NSTX in high harmonic fast wave heating and current drive. DIII-D is planning to schedule the ICRF activities in blocks of time to facilitate collaboration on the ICRF experiments and reduce the effort in maintaining that system to shift resources to the ECH system.
- C-Mod has de-emphasized disruption and detailed MHD stability studies untill the MSE system is installed and operational. The MSE diagnostic entails strong participation by their collaborators (PPPL and UT).
- DIII-D will emphasize high-beta wall stabilization experiments whereas C-Mod is not emphasizing high-beta operation.
- C-Mod cryopump is deferred and experiments on divertor pumping will be conducted on DIII-D which plans to add further panels at the end of FY'99.
- Studies of Alfven eigenmode instabilities will be emphasized on JET and JT-60U and de-emphasized on DIII-D and C-Mod.
- C-Mod has modified the baffling of the divertor to test the effect of neutral backstreaming. DIII-D earlier deferred the divertor modification till FY'98-00.

In some cases, coordination of activities entails the execution of complementary experiments either because these enable the study of the science over a broader range of parameters or to assess the generality of the result. Examples include:
- Study of the force balance on impurity flows in the scrapeoff region between facilities with differing divertor geometries.
- Dimensionless scaling experiments of confinement between facilities with different physical dimensions.
- Studies of internal transport barriers between facilities with different heating and momentum systems.
- Collaboration on JET and JT-60U enables us to extend our research to higher levels of performance and over a wider range of dimensionless parameters such as ρ^*. This has occurred in a number of areas including advanced tokamak studies and dimensionless scaling experiments.

One of the issues which was discussed was the continuation of the ITER Expert Groups which have facilitated data exchange and the creation of international databases. At this time, the frameworks for international collaboration are being reassessed. The consensus viewpoint was that our scientist want to be involved in international collaborations and exchanges and that a framework for assembling data and working together is needed and hopefully will evolve from the current ITER process. It was noted that the ITER Project provides a very useful focus to this activity but with the natural consequence of limiting attention given to broader and longer-range topics of importance. In any case, the present U.S. membership on the ITER Expert Groups should be revisited.

A topic, which came up late in the meeting, was the process for proposing significant upgrades to the facilities. The C-Mod group, in collaboration with PPPL, is proposing the installation of a lower hybrid current drive system and the DIII-D group is already upgrading the present ECH/ECCD system from 3 to 6 MW and at the end of CY'00 plans to assess whether to increase the power to 10 MW. (PPPL is also a collaborator on the DIII-D ECH/ECCD experiments.) It is important to note that the FFCC is not a review committee and will not make recommendations regarding upgrades. However, the Committee recognizes that a major element of the tokamak program is current profile control to achieve high-beta discharges. We recommend that a workshop be organized on advanced tokamaks with the focus on current profile control to sustain high beta plasmas and that it be held prior to the FTP meeting. This is a complex topic involving the integration of current profile control and evolution, MHD stability, divertor operation, and confinement. At that meeting, a discussion of the plans and projected research for both the C-Mod lower hybrid system and the DIII-D ECH/ECCD system should be given.

Plans for subsequent meetings:

DIII-D PAC	February 8-9
NSTX PAC	February 11-12
C-Mod	TBD

FFCC	February 25-26 at GA
FTP	April 7-9

Appendix A

Transport Control

By K. Burrell, M. Greewald, and E. Synakowski

-Advances in theory and modelling, innovations in experiments and diagnostics, and stronger coupling between these elements has resulted in vastly improved understanding of transport phenomena and striking improvements in plasma performance. The thrust of the transport program in the next two years will be to build on our improved physics understanding and to extend the improved confinement regimes into reactor relevant scenarios. The major US facilities have essential unique capabilities. DIII-D has great flexibility for plasma shaping, a wide variety of auxiliary systems, and an outstanding diagnostic set. C-Mod provides data in a unique regime of high field and density, with equilibrated ions and electrons, and with a heating source decoupled from the particle and momentum source. NSTX will provide definitive tests for physics at very low aspect ratio and should extend transport studies into a new regime of very strong shear and very high beta.

1. ***Tests of transport and turbulence models***

C-Mod- Transport and fluctuations: new diagnostics (BES, upgraded reflectometry, heterodyne ECE)
- Marginal stability tests w/ off-axis mode-conversion heating and cold pulses

DIII-D - Distinguish between theory-based transport models with explicitly designed experiments
- The fundamental nature of plasma turbulence (e.g. signatures of self-organized criticality)

International collaborations - Transport code benchmarking: US, ASDEX-U, Tore-Supra, JT-60U, with emphasis on size scaling

2. ***Core Transport Barrier Physics and Control***

C-Mod - RF flow drive using ICRF and mode converted IBW. Improved profiles with new DNB
- Intermachine study on ITB formation and sustainment: role of particle and momentum source

DIII-D - Regulate edge bootstrap current and/or pressure gradient to extend the duration of AT modes
- Develop target discharges for NCS AT plasma (ECH emphasis)
- Expand the spatial extend and time duration of internal transport barriers
- Study detailed physics of core transport barrier formation

NSTX - Initial tests of effects of small aspect ratio on E¥B shear and turbulence using RF

International collaborations - Size scaling for ITB formation (ASDEX-U, JET, JT-60U)
- PPPL/JET development of MSE system and reflectometry for fluctuation measurements
- RF-assisted internal transport barriers on JET will be investigated with US participation.
- Electron thermal transport will be studied using the ECRH capabilities on Tore Supra

3. ***H-mode Physics***

C-Mod- Pedestals: studies of width and height and correlation with relevant physics parameters
- Thresholds: local scalings; neutral, SOL, ∇B effects; compare to drift-Alfven turbulence models
- Basic physics of "small ELM" regime with detailed studies of Enhanced D-alpha H-mode
- Impurity transport studies including determination of transport coefficient profiles.
- Density limit and (H/L) transition

DIII-D - Improve experiment/theory comparisons- edge and divertor conditions needed for H-mode
- Search for means of lowering the L to H power threshold
- Study H-mode edge pedestal to determine key physics controlling gradients and pedestal values

NSTX - Study H-mode power threshold at low B field and small aspect ratio
- Assess edge pedestal characteristics at small aspect ratio, low B using Thomson scattering

International collaborations - Edge pedestal characteristics with ASDEX-U, JET and JT-60U
- Edge transport comparison between C-MOD and JT-60U
- Role of neutrals in the L-H transition will be explored on ASDEX-U and JT-60U.

4. ***Non-dimensional Transport Studies***

C-Mod -Dimensionless scaling of transport and turbulence: experiments with DIII-D, ASDEX, and JET)

DIII-D - Use dimensionless scaling approach to define an attractive next-step ELMing H mode device
- Detailed tests of nondimensional scaling with C-MOD and JET

NSTX - Initial aspect ratio scaling will begin in FY '00

International collaborations - JET size scaling studies. Proposed ASDEX-U/C-Mod similarity experiments

5. ***Impurity Effects on Transport (e.g. RI mode and impurity effects on turbulence)***

C-Mod - Some effort to look for RI behavior

DIII-D - RI mode experiments including turbulence studies. Edge regulation of bootstrap current and/or pressure gradient to extend AT modes

International collaborations - Collaborative study of RI mode physics on JET is planned

- Continue RI mode comparisons (TEXTOR and DIII-D, C Mod, ORNL)

Large Report on the High-beta technical area (Granetz, Manickam, Navratil and Strait)

High Beta Stability

By R. Granetz, T. Strait, J. Manickan

The overall goal of high-beta steady state operation requires success in five scientific topical areas: Disruptions, Beta-limiting MHD activity, Profile control, Localized MHD activity and the interaction of energetic particles with MHD modes. In each area strategies for both understanding and control are required. The proposed program of activities in the US fusion facilities addresses all these areas with varying degrees of emphasis. It should noted that the activities on most of the facilities will occur during both FY'99 and FY'00, however in FY'99 NSTX will be in a start-up phase and the activities listed for it are largely expected during the second year. There is also a significant design effort for a quasi-axisymmetric device which is currently nearing completion and may be featured in a future report. Finally it should be noted that international collaborations also have a significant overlap with many of these topical research areas.

C-Mod will determine the disruption forces that the inner divertor must accommodate in order to operate at currents in excess of 2.0 MA. DIII-D will demonstrate disruption mitigation by gas and pellet techniques, with the goal of determining the need for cryogenic liquid jets. NSTX will measure halo currents and begin characterization of disruptions in STs.

Investigation of beta-limiting instabilities will focus on neoclassical tearing modes (NTM) and resistive wall modes. With the addition of 4 MW of ICRF power, C-Mod will explore regimes of higher beta. DIII-D goals are to validate neoclassical tearing theories in order to project to future machines, and investigate stabilization of NTMs with ECCD. Resistive wall mode (RWM) experiments in DIII-D will determine the required plasma rotation and wall separation for passive stabilization of n=1 kink modes by a resistive wall, and begin experiments to improve stabilization with active non-axisymmetric coils. HBT-EP will determine the optimum wall configuration for RWM stabilization and study feedback control. NSTX will identify the MHD activity which determines beta limits in low aspect ratio plasmas.

Current profile control is discussed in the parallel report on 'Current Drive Physics'. Here we only describe some related issues which also impact on profile control. There are no direct pressure profile control experiments.

CMOD will make direct measurement of J(r) with a new DNB MSE which will help AT/RS operation. NSTX will determine the efficacy of non-inductive start-up and CHI.

The focus here is on sawteeth, NTMs and ELMs. Both DIII-D and C-Mod will investigate the physics of sawtooth stabilization by non-inductive current profile control (ECCD in DIII-D, and ICRF in both machines) and by ICRF-driven fast ions. Stabilizing sawteeth may also have an effect on NTMs by eliminating a source for seed islands. In addition DIII-D will attempt direct stabilization of NTMs. Both C-Mod and DIII-D will advance the physics understanding of the instabilities driven by the H-mode edge pedestal. DIII-D experiments will also aim at controlling ELMs by modifying the edge gradients, collisionality, or stability limits. NSTX will identify the role of localized MHD activity especially during a fast current ramp up.

Both C-MOD and DIII-D will demonstrate long-pulse sawtooth stabilization with ICRF heating. NSTX will investigate TAE modes at low aspect ratio.

Non-inductive Current Drive

By C. K. Phillips, R. Prater, and S. Wukitch

A balanced and coordinated program of scientific research has been generated to address the key issues for the physics of current drive and its applications. The current drive techniques are selected to be compatible with the unique characteristics of these facilities. Alcator C-Mod is a tokamak with high toroidal magnetic field which will utilize current drive in the ICRF and lower hybrid regimes. DIII-D is a tokamak with moderate toroidal field for which ECCD is the primary approach, with FWCD and NBCD playing an auxiliary role. NSTX is a low field spherical torus, where high harmonic FWCD, helicity injection, and NBCD are appropriate. The physics and technology of these current drive techniques are uniquely matched to the magnetic field and the programmatic thrusts of each device. The science goals of the current drive program include the following:

- Develop full predictive capability and experimental verification of current drive for radiofrequency power in the ion cyclotron (FWCD and MCCD), lower hybrid (LHCD), and electron cyclotron (ECCD) regimes and for neutral injection (NBCD). The physics for rf waves and for neutral particle beams includes propagation, absorption, distortion of distribution functions, parasitic losses, and nonlinear effects, as a function of plasma density, temperature, beta, aspect ratio, geometry, fast particle population, and power. In FY99-00, experiments on DIII-D and C-Mod will address issues pertaining to ECCD and MCCD, respectively, for comparison to code calculations. Measurements of ICRF fields in C-Mod and DIII-D will test theories of wave propagation. On NSTX, experiments to test theories of high harmonic fast wave (HHFW) physics will begin in FY99, with experiments combining NBCD to start late in FY00. Modeling is being advanced in all programs to include rf-driven currents self-consistently in the equilibrium. Initial experiments to test the efficiency of coaxial helicity injection (CHI) for current drive will also commence on NSTX in FY99.
- Demonstrate and understand the role of rf power in driving poloidal rotational shear for generation and control of internal transport barriers and pressure gradients, leading to sustainment of tokamak discharges with high confinement and high beta. Experiments should test the effect of localized CD, localized momentum input, and highly localized heating with its effect on the Reynold's stress, as means to generate the flow shear. A substantial part of the experimental programs on C-Mod and DIII-D will be addressing these questions, using ICRF on C-Mod and ECH/ECCD and ICRF on DIII-D. Synergism between HHFW heating and the self-induced sheared poloidal flows predicted to be present in NSTX discharges will be studied.
- Develop discharges sustained fully noninductively by current drive, with high bootstrap fraction and with real time control over the profile of driven current. Major issues include current diffusion and profile relaxation, the behavior and control of discharges with high bootstrap fraction, and development of means to align the bootstrap current and the driven current with the desired current. On C-Mod, MCCD for off-axis current drive with a new antenna will begin, and preparations for LHCD will continue. On DIII-D, experiments will test the modeled scenarios for discharges with high normalized performance using ECCD to sustain the current profile. Initial tests of HHFW current drive will begin after a year of NSTX operation.
- Develop fully noninductive startup and current ramp techniques in preparation for an attractive low aspect ratio torus without an ohmic heating coil. This is an important topic for reactors, particularly those based on the spherical torus concept, since elimination of the central leg of the OH transformer significantly improves the concept. Experiments planned for NSTX over the next two years will test helicity injection for this purpose. DIII-D will use ECH breakdown followed by bootstrap current overdrive to demonstrate startup.
- Develop techniques to suppress internal MHD instabilities such as sawteeth and neoclassical tearing modes and to generate toroidal rotation for suppression of resistive wall modes and other external MHD activity. Recent results >from the world tokamak community indicate that these are key instabilities limiting performance in advanced tokamaks. Experiments will begin on DIII-D using ECCD to stabilized neoclassical tearing modes and to determine the effects of high power heating (FWCD, ECCD, and NBCD) on toroidal and poloidal rotation. The effects of energetic particles on MHD will also be addressed. MCCD will be used on C-Mod for these same purposes. Mechanisms for the observed bulk toroidal rotation observed on C-Mod with ICRF heating and its viability for the suppression of global MHD modes will be examined.
- Develop the technology needed for the various schemes that will yield efficient coupling and robust real time control of the current profile. This work is being carried out by all three laboratories. NSTX will study the requirements on launcher design placed by the effects of highly sheared edge magnetic fields. Advanced launchers which allow for real-time plasma control will be designed for the LHCD system of C-Mod and the ECCD system of DIII-D.

Power and Particle Exhaust

By S. Allen, B. Lipschultz, and D. Stotler

The ultimate objective in this area is to safely disperse the **power exhaust** from the core plasma while controlling its density, maximizing energy confinement and keeping impurity levels low. Success will require an understanding of the intrinsic physical processes and their synergism's. This knowledge will be applicable to future devices, tokamaks as well as alternate concepts. The flows of particles, energy, and momentum between the confined plasma and the surrounding material surfaces are governed by parallel and perpendicular **transport** mechanisms. Particular attention is given to **impurity transport** (including He) because of the need to minimize core radiation and fuel dilution, while maximizing radiation in the boundary and divertor plasmas. The generation of impurities, unwanted retention of hydrogen isotopes, and tritium safety concerns are addressed by the study of **plasma surface interactions** and conditioning. All of these physical processes are dependent on external conditions such as geometry (divertor plasmas shape), choice of first-wall material (high- or low-Z) and conditioning techniques.

Radiation is the primary tool utilized to dissipate the glow of power flowing out of the core. One implementation of this approach, the detached divertor is being studied regarding the role of the private flux zone (DIII-D), the flows of ions and neutrals (DIII-D and C-Mod), and how deuterium atomic physics affects particle, momentum and energy transport (C-Mod). The RI radiating boundary layer mode is being studied in neutral beam heated plasmas (DIII-D) and ICRF-heated plasmas (C-Mod) to understand the role of particle and momentum source.

The understanding of transport coefficients in the plasma boundary is at a primitive level. DIII-D brings to bear the very detailed 2-D measurements of n_e, T_e, and Mach number in comparisons with the plasma modeling code UEDGE-DEGAS. C-Mod research takes a more empirical approach using a TRANSP-like interpretive analysis to achieve scalings over a large number of discharges. The underlying processes driving this transport will be studied with SOL Langmuir probe (DIII-D) and visible light (C-Mod) turbulence measurements. NSTX boundary characterization will help determine if there are issues related to the aspect ratio.

Impurities are often the limiting factor in all confinement schemes. Erosion and redeposition processes at the tile surface will be studied by the DiMES probe (DIII-D) and with integration markers on Mo tiles in C-Mod. JET ellipsometry provide real-time tile erosion measurements. The transport of the impurity ions, which escape the near-tile region, towards the core is being characterized in DIII-D with/without pumping and in C-Mod with mass-dependence experiments. Active methods of controlling impurity transport are being pursued by inducing flows utilizing simultaneous puff and pumping in DIII-D and through recirculation of neutrals in C-Mod.

Some of the geometrical choices made for these experiments can have important effects on the plasma. Variations in divertor geometry continue to be pursued with comparisons between the gas-target of JET, C-Mod's vertical plates, and the variable divertor length of DIII-D. The comparisons of low (JET), medium (C-Mod) and high (DIII-D) triangularity carried out in core transport studies lead to corresponding investigations of the effect on boundary characteristics. Single-null plasmas (C-Mod, DIII-D, JET, NSTX) potentially provide greater field line length for the dissipation of power flows, while a double null (DIII-D, NSTX) increases the strike point area. NSTX will provide input on the effect of low aspect ratio and inner-wall limited operation.

We will continue to optimize the operation with graphite tiles in NSTX_ and DIII-D due to its heat flux handling capability. High-Z materials are more reactor-relevant because of T-retention and neutron damage. We will pursue this type of first-wall operation with Mo in C-Mod. The erosion and redeposition characteristics of these materials differ and need to be compared. Specific data on the retention and recovery of tritium in graphite will be examined on JET. (Left out boronization.). Cleaning techniques applicable with magnetic fields (electron-cyclotron - C-Mod) and without (DIII-D) are being developed.

Integration of Improvements
By D. Baker, P. Bonoli , and S. Kaye

AT Issues

One of the primary goals of the magnetic fusion energy research program is to provide a scientific basis for the design of a fusion based energy production facility. This will require an integration of research in a wide range of topical areas such as transport, stability, current drive and divertors. The Advanced Tokamak (AT) research program is a major aspect of this integration. At each facility, DIII-D, C-Mod and NSTX, integrated AT programs exist which will address the effort to combine good confinement, high stability, long pulse non-inductive current and efficient particle and heat exhaust. Each facility will utilize its unique features to address the AT effort from different approaches. For example in the coming year: DIII-D will utilize its 3 MW ECH and ECCD system combined with a pressure profile with high bootstrap fraction to produce a plasma with close to zero inductive current, a normalized beta of about 4 and a confinement factor which in projected to be between 2 an 3. C-Mod, with short current penetration times relative to the pulse length, will investigate non-transient ITB formation using off-axis mode conversion electron heating and current drive. NSTX, in its first two years of operation, will investigate CHI for plasma initiation and RF assist to generate sheared flow for ITB formation and RF current drive for long-pulse operation. The experimental plans of the three facilities which relates to AT research is listed below.

A) Demonstration of non-transient optimized shear at high- β_n (high bootstrap fraction): C-Mod, with short current penetration times relative to the pulse length, will investigate non-transient ITB formation using off-axis mode conversion electron heating and current drive. DIII-D plans a series of "counter beam injection" experiments to study EXB shear stabilization with evolving q profiles. NSTX will investigate RF assist to generate sheared flow for ITB formation, RF current drive for long-pulse starting in FY99 and continuing with higher power and profile diagnostics in FY00. Pellets may be a possibility in FY00. PPPL will participate in experiments with DIII-D, JET, and JT-60U.

B) Stability of a good confinement, edge pedestal with high bootstrap fraction: DIII-D will attempt to regulate (suppress) the edge bootstrap current and/or the edge pressure gradient to extend the duration of AT modes using a variety of techniques including impurities, gas puffing pumping, edge shaping and small pellets. DIII-D will also investigate the neoclassical tearing model and attempt to stabilize this mode with ECCD. NSTX will investigate, primarily in FY00, the role of neoclassical tearing and the effect of wall stabilization aided by RF induced rotation in these configurations. PPPL personnel will participate in experiments on DIII-D, JT-60U, and C-Mod.

C) Transformerless operation: NSTX will investigate CHI for plasma initiation starting in FY99, with bootstrap overdrive scenarios, and the role of neoclassical tearing modes, possible for FY00. HHFW and high bootstrap fractions will be studied for current sustainment starting in FY00. PPPL will participate in FW current drive experiments on C-Mod. DIII-D will investigate HI startup, Outer F-coil startup, and plasma current rampup and steady-state operation using bootstrap and non-inductive current drive.

D) High density ($n_e>n_{GW}$), high confinement operation: C-Mod plans a series of experiments to test density limits. DIII-D has an ongoing collaboration with TEXTOR in the area of RI-modes. RI-modes present a good opportunity for research on larger devices with U.S. collaboration. NSTX will possibly investigate high density operation in FY00 with pellets.

E) Long pulse, high power density: C-Mod will explore detached radiative divertor physics in FY99, and will compare results with those of DIII-D. C-Mod will also explore RI-modes, in collaboration with DIII-D. DIII-D will develop a target plasma for high beta (βn ~ 4), high confinement plasma (H ~ 3) with close to 100% non-inductive current using EC (in 1999, ~ 3MW) current drive. This topic will be explored on NSTX in FY00 with high power HHFW current drive and heating. Tore-Supra and TEXTOR (RI-modes) present opportunities for collaboration.

F) Low disruptivity, reduced severity (high-β, long-pulse): NSTX and PPPL collaborators on MAST will investigate disruption severity in ST through halo current measurements over a range of operational space.

Wall stabilization will be utilized in NSTX. PPPL will collaborate with DIII-D on installing saddle coils and other hardware for detection and control. C-Mod will examine structural forces on inboard wall/divertor region, where the halo currents are the greatest to determine if strengthening is required for higher current operation. DIII-D will study the wall stabilization of high beta, and disruption using "killer" pellets and massive gas injection.

G) Plasma control for advanced scenarios: NSTX, in collaboration with DIII-D, will be developing a plasma response model and implementing rtEFIT. This is the first step towards a real-time determination of plasma stability and profile feedback control. Opportunities for collaboration exist on Tore-Supra.

H) Computer modeling of combined scenarios to develop predictive capability: Some GK/GF work on NSTX operational scenarios will be carried out in FY99. C-Mod will utilize ACCOME, coupled with JSOLVER and PEST-II, to analyze the stability of current driven AT scenarios. Opportunities exist for modeling and benchmarking AT results on JET, DIII-D, and JT-60U.

I) Compact systems: NSTX and PPPL collaborators with MAST will assess the ST development path.

Burning Plasma Physics Issues

A) DT confinement and heating: PPPL collaboration with JET in understanding thresholds for transition boundaries (internal and edge), the source of the isotope scaling (core vs. pedestal), and α -heating. DIII-D intends to participate in JET DT experiments on optimized shear and hot ion modes.

B) He transport studies in the core and SOL: Opportunities exist for PPPL collaboration on JET DT experiments and AT plasmas on DIII-D and JT-60U.

C) α/fast particle effects

1) Collective effects (super-Alfvenic): PPPL will participate in TAE mode studies on JET and JT-60U. These studies will also be carried out in NSTX when the neutral beam $(v_b/v_\alpha \geq 1)$ comes on line in FY00.

2) Stabilization of neoclassical tearing modes and ELMs: Opportunities for collaboration exist on JET, JT-60U (NINB), and ASDEX-U.

3) RF/fast ion interactions: NSTX will investigate NBI/HHFW interactions, the effect on heating and current drive efficiencies, and the possible loss of fast particles starting in FY00. PPPL will participate in NINB/LHCD experiments on JT-60U. Opportunities for collaboration exist in JET DT experiments.

D) Current ramp-up: C-Mod will explore confinement, density control, and operating limits during the initial current rise. This is relevant for igniting plasmas.

APPENDIX B

Plasma Transport

By R. Hazeltine

Plasma transport---the diffusion of heat and mass across the confining magnetic field---is the fundamental issue for confinement of a macroscopically stable plasma. Historically plasma transport has been dominated by the almost universal presence of turbulence; indeed, the electromagnetic interaction has made the plasma medium ideal for studies of turbulence, allowing detailed diagnosis and even control. The discovery of high-confinement operating regimes, in which the plasma is quiescent and transport approaches the irreducible minimum required by Coulomb collisions, has revolutionized expectations of plasma confinement, affecting the design and operating characteristics of proposed fusion reactors. It has also brought new excitement to the broader science of fluid turbulence.

The beginnings of a comprehensive picture of high-confinement regimes, their associated "transport barriers," and the crucial roles of velocity and magnetic field shear in quieting turbulence, is emerging. A major focus of present research is the consolidation of this picture and its detailed confirmation. The goal is not only a controllable handle on plasma confinement, but also progress on one of the historical challenges of physics: to understand and control turbulence.

A second focus concerns the tokamak edge region, where plasma interacts with vessel walls. In particular one seeks improved understanding of the tokamak divertor, a magneto-mechanical structure that provides control over plasma heat and particle losses. In this area transport *parallel* to the magnetic field, along with plasma-neutral interactions and atomic physics effects, plays the key role. Classical parallel transport theory, based on short mean-free path theory, does not resolve the problem, mainly because the tokamak edge region includes subregions in which the mean-free path exceeds gradient scale lengths.

Examples of major goals for FY99-FY'00 are:
Establish parametric dependence of edge pedestal in H (high-confinement)-mode. An emerging consensus in the community is that plasma physics at the edge has a profound influence over core confinement, and therefore over accessibility of ignition. The temperature and density just inside the edge transport barrier are governed by plasma stability and transport. Determining the dependence of these pedestal characteristics on other plasma parameters will allow more reliable understanding of fusion ignition requirements, more detailed comparisons to stability predictions and better general understanding of high-confinement physics.

Develop the data base for the reliable application of velocity and magnetic-field shear in turbulence suppression. Few breakthroughs in fusion science have brought more excitement---in terms of both fusion reactor design as well as general scientific importance---than the ability to suppress turbulence. The detailed experimental and theoretical studies that exploit and consolidate this advance are therefore a high priority. Previous work has shown that velocity shear plays the critical role in barrier formation; its relation to (reversed) magnetic field shear, whose presence encourages barrier formation in the plasma core, remains a puzzle---tantalizing because of the confinement improvements that would be likely to come with more complete understanding. Planned experimental attacks on this issue should lead to more reliable control of transport in future confinement experiments.

Test theoretical models for plasma transport parallel to the magnetic field outside the separatrix, including the long mean-free path effects. The heat load on various plasma-facing components, such as the divertor collector plate, depend upon the variation of plasma temperature and density parallel to the magnetic field in the edge region outside the divertor separatrix. In addition to the complex atomic physics and plasma-neutral issues affecting this region, the nature of parallel transport plays a crucial role. Since parts of the region have steep gradients but weak Coulomb collisions, long mean-free path physics enters as importantly here as it does in studies of the earth's magnetosphere. Studies of parameter variation along the field would test theoretical descriptions of long mean-free path effects, and allow better predictions regarding such phenomena as plasma detachment from the divertor plate.

High Beta Stability

By G. Navratil

The fusion power density achievable in a power plant depends fundamentally on the level of plasma pressure which can be confined by the magnetic configuration while maintaining MHD stability. The achievement of such high-beta plasma states, where beta is the ratio of plasma pressure to magnetic field pressure, is a crucial factor in the overall fusion power plant economics. Because of this central role played by high beta plasmas in fusion energy research, advances in MHD theory, including the effects of finite electrical conductivity, to predict achievable beta limits has been a driving force in fusion energy science. These advances provide the basis for large scale scientific computation using two and three dimensional equilibrium and stability codes. Optimization of the stable beta limit has been the design basis for experimental tests of most magnetically confined fusion concepts.

Progress in our understanding of the behavior of high beta fusion plasmas in the laboratory is also of broader interest in the development of plasma physics. Most astrophysical plasmas near stars and planetary magnetospheres have relatively high beta values. Advances in MHD theory and supporting experiments carried out for fusion energy research, have provided the framework for analysis of important astrophysical MHD phenomena including magnetic reconnection, anomalous plasma heating, turbulence and dynamo phenomena, and the evolution of magnetic field topology. Important insights have been gained on long-standing problems of solar coronal heating and the magnetic reconnection dynamics in the earth's magnetotail.

In our largest tokamak experimental facilities, recent advances in the capability to make accurate measurements of the magnetic field distribution inside a hot fusion plasma using the motional Stark effect, allow the theoretically predicted MHD stability limits for long wavelength MHD modes in tokamak experiments to be accurately determined. This improved quantitative understanding of beta limiting instabilities in tokamak plasmas predicted by ideal and resistive MHD theory is now becoming well established. The objective of research in this area of high beta stability in the next few years is to apply that understanding to actively control the onset of MHD instabilities to increase the operating beta limits in toroidal plasmas.

Examples of major goals for FY99-FY00 are:

Validate neoclassical tearing theory and test active control of Neoclassical Tearing Modes with Electron Cyclotron Current Drive. An important discovery in the past two years is the role of the self-generated bootstrap current, which is predicted in high beta plasmas by neoclassical theory, in the destabilization of tearing modes in toroidal plasmas. These tearing modes create magnetic island structures through the process of magnetic reconnection resulting in increased particle and energy losses. The use of local electrical currents driven by radiofrequency waves near the electron cyclotron frequency, has been predicted to stabilize these Neoclassical Tearing Modes. Experiments to study these effects on DIII-D will provide important insights in to the validation of the theory as well as provide a test of a promising technique to control the growth of these modes in a fusion power plant.

Validate the model of resistive wall stabilization of kink modes in a rotating plasma. It has been a long-standing prediction of MHD theory, that long wavelength, external MHD kink modes driven unstable at high beta, can be stabilized by the presence of a perfectly conducting wall near the plasma boundary. However, since realistic fusion power plant designs will necessarily have walls with finite electrical conductivity, magnetic field can diffuse through the conductor and the theory predicts these unstable MHD modes will have slowed growth rates but will not be fully stabilized. More recent theoretical studies have predicted that toroidal plasma rotation with a fixed normal conductor wall can stabilize these modes, and experimental evidence now exists to support that prediction. Experiments are planned on DIII-D, to quantify these stabilizing effects of rotation and the internal plasma dissipation mechanisms which are the basis for the predictions of rotational stabilization.

Confirm model of active feedback control of kink modes. In addition to the passive stabilization of kink provided by toroidal rotation with a conducting wall, these modes may also be stabilized by using active feedback control to simulate the response of a perfectly conducting wall. Active feedback control of the axisymmetric, n=0, mode is now routinely carried out on all tokamaks with non-circular cross-sections. Experiments and supporting 3 dimensional stability models will explore extending our capability for active mode control to the n=1 kink mode on DIII-D and HBT-EP in the next 2 years.

Wave-particle Interactions: Non-inductive Current Drive

By G. Navratil

One of the principal tools to affect a plasma in the laboratory is through the interaction of waves with the charged particles of the plasma. For many years plasmas have been created and sustained through application of radiofrequency power for heating the charged particles in the plasma. A wide range of detailed measurements of local plasma parameters (*e.g.* T_e, n_e, ϕ, and B) have also been devised by analysis of the scattered electromagnetic waves from the particles in the plasma. Both of these areas remain active subjects of plasma physics research. Today, a key scientific goal of fusion energy science is the extension of the plasma lifetime in high performance fusion plasmas from the transient duration of a few seconds towards a basis for true steady-state operation. This has focused considerable attention on another important area of application of wave-particle interactions in the non-inductive sustainment in steady-state of electric current flow in a fusion plasma.

The capability to control and sustain the distribution of electric current flow in a magnetically confined plasma, plays a critical role in fusion energy research since the plasma current distribution determines the equilibrium and stability properties of the magnetic confinement geometry. Radiofrequency waves are used not only to shape the equilibrium current distribution in a toroidal plasma to optimize the stable beta, but also are now predicted to induce sheared plasma rotation for control of plasma turbulence. In addition, MHD dynamo processes have been well established in toroidal plasmas and the use of MHD turbulence to drive current in the plasma through the injection of magnetic helicity is ready to be tested in larger scale, collisionless plasmas.

In FY99-FY00 examples of major goals in this area are:

Validate theoretical models and quantify the efficiency of off-axis current drive. In a tokamak the leading approach to steady-state, high beta plasmas for fusion power plants uses a configuration where a significant fraction (80% - 90%) of the plasma current is self-generated by the bootstrap current, but these plasmas still rely of a modest fraction of current needed to optimize the equilibrium to come from some other non-inductive source to drive current away from the magnetic axis. Several candidates have been identified to provide this current which are distinguished by the frequency range of the applied radio frequency energy launched into the plasma. In the next two years tests will be carried out using radio frequency power in the ion cyclotron frequency range on C-Mod and DIII-D, on high harmonics of the ion cyclotron frequency in NSTX, and with harmonics of the electron cyclotron frequency in DIII-D. The important questions which need to be resolved include the degree of spatial localization and control of the current distribution as well as the efficiency with which the current is driven.

Establish the role of radiofrequency power in the generation of poloidal plasma rotation shear flow for control of plasma turbulence. One of the exciting predictions to come out of the discovery of the role of sheared poloidal plasma flow in the suppression of plasma turbulence and its associated enhanced levels of heat and particle transport a few years ago, is that radiofrequency power applied to the plasma could be used to generate sheared rotational plasma flow in well defined local regions of the plasma. If successfully demonstrated in the laboratory, this would provide a powerful new tool to use in optimizing the plasma core of a fusion power system for both improved confinement and a stable higher beta. In the next 2 years, experiments to test these ideas will be carried out on C-Mod and DIII-D.

Demonstrate current drive in a collisionless toroidal plasma through the external application of magnetic helicity. One of the exciting new approaches to improving the tokamak concept has come from experiments conducted at very low aspect ratio (a/R < 1.5) which have confirmed theoretical predictions of stable, high beta operation much higher than conventional aspect ratio tokamaks. However, a key obstacle to the operation of these new class of low aspect ratio devices is the very small region near the major toroidal axis for the usual inductive transformer coil to initiate and sustain the confining plasma current. Exploratory experiments have shown success with the injection of magnetic helicity by application of an electric field by electrodes across the open magnetic field lines surrounding the closed flux surfaces confining the plasma. The NSTX experiment which begins operation in FY99, will carry out proof of principle experiments to test this concept in a collisionless plasma at the level of several hundred kA.

Power and Particle Exhaust

By D. Hill

The ultimate objective in this area is to safely disperse the power exhaust from the core plasma while controlling its density, maximizing energy confinement, and keeping impurity levels low. At present, experiments have shown that active control of the divertor heat flux is possible, but a number of issues are still outstanding which limit our confidence in designing efficient divertors. Further work to develop and test useful Success will require an understanding of the intrinsic physical processes and their synergism's is required. This knowledge will be applicable to future devices, a wide variety of magnetic fusion concepts, laboratory, and astrophysical plasmas.

During the past decade we have developed new insight into the relationship between atomic physics and basic plasma processes such as energy and momentum transport on open field lines. As a result, our capability to simulate numerically the tokamak divertor has improved dramatically, and codes can now predict some of the key properties of dense, highly radiating plasmas. A major of focus of research on power and particle exhaust lies in pushing comparisons between simulation and experiment to higher accuracy, clarifying the physics which governs plasma transport and plasma-surface interactions. At the same time, these results are being exploited to demonstrate reliable, active control of divertor conditions. Both of these aspects of boundary physics research are contained in three major near term objectives for power and particle control research:

Validate comprehensive numerical simulations of radiative divertors in tokamaks. The dynamics of the highly radiating divertor plasma result from the complex, nonlinear interaction of a number of physical processes including ionization, excitation, recombination, elastic scattering, molecular dissociation, and turbulent transport. Integrating these effects to produce an understanding of the macroscopic behavior of the edge and divertor plasma requires sophisticated 2-d numerical simulation codes which can accurately account for the magnetic topology, as well as the geometry and composition of first wall surfaces. These codes now reproduce many features observed in highly radiating detached plasmas (defined by a strong reduction in plasma pressure on open field lines). Further detailed measurements of the 2-d structure of plasma density, temperature, radiation, and ion/neutral flow patterns under a variety of conditions are needed to fully test these models.

Control impurity transport in the edge plasma through the use of friction to counteract ion thermal gradient forces on open field lines. The balance between these forces is thought to be a determining factor governing our ability to enhance divertor impurity radiation while maintaining a clean core plasma (Enhanced divertor radiation is needed to reduce the peak divertor heat flux in a tokamak) Neutral transport effects and the tokamak geometry may limit the importance of the ion dynamics. Experiments will attempt to correlate the measured ion flow to measured impurity enrichment.

Measure turbulence-driven transport coefficients on open field lines near the magnetic separatrix. The radial transport coefficients set the radial scale length for the plasma on the open field lines outside the magnetic separatrix and so directly govern the plasma properties (density and temperature) in the divertor region. The relationship between these transport coefficients and the fundamental characteristics of the edge plasma (e.g., collisionality, beta) are poorly understood, limiting our ability to predict important divertor plasma parameters for a given configuration. Edge measurements will build a database that can be used to test turbulent transport models.
The research conducted in these three areas addresses fundamental questions relating to the flows of particles, energy, and momentum on open field lines and as such, is directly applicable to other physical systems such as the earth's magnetosphere, the solar corona, and the laboratory-sized plasmas used in semiconductor production. In addition, studies of the plasma surface interaction at the divertor targets provides important data for the surface physics community which, in turn, develops the models for physical sputtering and chemical erosion that play such an important role in material selection for fusion experiments.

Integration of Innovations

By R. J. Hawryluk
Research on the major facilities develops both the underlying understanding and the innovations required for fusion energy development. The research objectives highlighted in the previous sections provide examples of this. An important element in the development of fusion energy is to incorporate promising new innovations and assess their impact. This is done both by system code analysis which evaluate the impact on cost and performance of future devices and by front-line experiments which not only benchmark our codes but also evaluate the interplay between various physics topics such as transport and MHD stability.

Develop the basis for choosing single vs. double null and the optimum triangularity for AT plasmas. A major issue in the design of a lower cost ignition device is the shape of the plasma and which must be chosen early during the design phase. The shape of the plasma has significant ramifications for the maximum pressure that can be sustained, the stability of the edge region, and the power and particle exhaust. DIII-D has the capability to assess the differences between single and double null operation and to vary the triangularity of the plasma. Though previous studies of discharges with monotonic q-profiles have been conducted and extensively utilized in machine designs, the effect of plasma shaping on AT plasmas with reversed shear and internal transport barriers has not been sufficiently characterized and understood. This work will be completed during 1999.

Begin operation of NSTX, a very low-aspect ratio spherical torus. This new proof-of-principle facility will extend the studies to strong plasma shaping. Construction is underway and experimental research will begin in FY'99.

Characterize the confinement and stability of optimized shear discharges in lower ρ^* regimes by collaborating on JET and JT-60U. Though there are many similarities between the results from DIII-D and TFTR on reversed shear discharges with internal transport barriers, there are some significant outstanding differences which can be attributed to current profile, plasma shaping, edge transport barrier, and ρ^* among other parameters. Nonetheless, the results from DIII-D and TFTR together with system studies have identified promising new directions for tokamak research. The recent results on JET and JT-60U have already demonstrated good confinement and fusion power production but have not achieved the good MHD stability which was achieved transiently on DIII-D for plasmas with similar shape. Understanding the interplay between transport and MHD stability in these larger devices will enable us to extend our results and reduce the extrapolation to future burning plasma advanced tokamak devices. Results on this will be available in FY99.

Develop a high beta (β_N=4), high confinement plasma (H ~ 3) with near fully non-inductive current drive for five seconds. High performance advanced tokamak discharges have ~~not~~ been sustained in present experiments for one second. Modeling indicates that controlling the evolution of the current profile such as by means of electron cyclotron current drive can extend the duration of high performance. This will be tested in FY01 with a 6 MW system on DIII-D to verify the models and improve our ability to predict the performance of future devices. DIII-D will work toward this goal in FY99 & 00 with a 3 MW short pulse ECH system.